UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 7, 2009

Mr. David A. Christian
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION, UNIT NO. 1, ISSUANCE OF AMENDMENT REGARDING MODIFIED INTERIM ALTERNATE REPAIR CRITERIA FOR B STEAM GENERATOR TUBE REPAIR (TAC NO. ME1191)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 264 to Renewed Facility Operating License No. DPR-32 for the Surry Power Station, Unit No. 1. The amendment changes the Technical Specifications (TSs) in response to your application dated May 5, 2009, as supplemented by letter dated May 6, 2009.

The proposed amendment modifies the Unit 1 interim alternate repair criteria (IARC) in TS 6.4.Q, "Steam Generator (SG) Program," and TS 6.6.3, "Steam Generator Tube Inspection Report." The proposed emergency license amendment request incorporates a modified IARC for the Unit 1 B SG tube repair criteria during Unit 1 Refueling Outage 22 and the subsequent operating cycle and revises the reporting requirements for the Unit 1 B SG.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

for

John Stang, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-280

Enclosures:
1. Amendment No. 264 to DPR-32
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 264
Renewed License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated May 5, 2009, as supplemented by letter dated May 6, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Renewed Facility Operating License No. DPR-32 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 264, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert Martin

Melanie C. Wong, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. DPR-32
and the Technical Specifications

Date of Issuance: May 7, 2009

ATTACHMENT

TO LICENSE AMENDMENT NO. 264

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

Replace the following pages of the License and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
License	License
License No. DPR-32, page 3	License No. DPR-32, page 3
TSs	TSs
3.1-13	3.1-13
3.1-14a.	3.1-14a
3.1-14b	3.1-14b
4.13-1	4.13-1
4.13-2	4.13-2
-	4.13-2a
6.4-13	6.4-13
6.6-3a	6.6-3a
-	6.6-3b

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. Maximum Power Level

      The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2546 megawatts (thermal).

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 264, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

   C. Reports

      The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

   D. Records

      The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

   E. Deleted by Amendment 65

   F. Deleted by Amendment 71

   G. Deleted by Amendment 227

   H. Deleted by Amendment 227

   I. Fire Protection

      The licensee shall implement and maintain in effect the provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report and as approved in the SER dated September 19, 1979, (and Supplements dated May 29, 1980, October 9, 1980, December 18, 1980, February 13, 1981, December 4, 1981, April 27, 1982, November 18, 1982, January 17, 1984, February 25, 1988, and

C. RCS Operational LEAKAGE

Applicability

The following specifications are applicable to RCS operational LEAKAGE whenever Tavg (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit).

Specifications

1. RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 1 gpm unidentified LEAKAGE,

c. 10 gpm identified LEAKAGE, and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG), with the following exception. The primary to secondary LEAKAGE for the Unit 1 B steam generator will be limited to 20 gallons per day during Operating Cycle 23.

2.a. If RCS operational LEAKAGE is not within the limits of 3.1.C.1 for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE, reduce LEAKAGE to within the specified limits within 4 hours.

b. If the LEAKAGE is not reduced to within the specified limits within 4 hours, the unit shall be brought to HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

3. If RCS pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within the limit specified in 3.1.C.1.d, the unit shall be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES - Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is 1 gpm or increases to 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis. Due to the permeability variation indications in the Unit 1 B steam generator found during Refueling Outage 22, the primary to secondary leak rate for that steam generator is limited to 20 gallons per day for Operating Cycle 23.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a main steam line break (MSLB) accident. Other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR (Ref. 2) analysis for SGTR assumes the contaminated secondary fluid is released via power operated relief valves or safety valves. The source term in the primary system coolant is transported to the affected (ruptured) steam generator by the break flow. The affected steam generator discharges steam to the environment for 30 minutes until the generator is manually isolated. The 1 gpm primary to secondary LEAKAGE transports the source term to the unaffected steam generators. Releases continue through the unaffected steam generators until the Residual Heat Removal System is placed in service.

The MSLB is less limiting for site radiation releases than the SGTR. The safety analysis for the MSLB accident assumes 1 gpm total primary to secondary LEAKAGE, including 500 gpd leakage into the faulted generator. The dose consequences resulting from the MSLB and the SGTR accidents are within the limits defined in the plant licensing basis.

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LIMITING CONDITIONS FOR OPERATION - RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

   No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 3). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures. Due to the permeability variation indications in the Unit 1 B steam generator found during Refueling Outage 22, the primary to secondary leak rate for that steam generator is limited to 20 gallons per day for Operating Cycle 23.

APPLICABILITY - In REACTOR OPERATION conditions where T_{avg} exceeds 200°F, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In COLD SHUTDOWN and REFUELING SHUTDOWN, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.1.C.5 measures leakage through each individual pressure isolation valve (PIV) and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

4.13 RCS OPERATIONAL LEAKAGE

Applicability

The following specifications are applicable to RCS operational LEAKAGE whenever T_{avg} (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit).

Objective

To verify that RCS operational LEAKAGE is maintained within the allowable limits.

Specifications

A. Verify RCS operational LEAKAGE is within the limits specified in TS 3.1.C by performance of RCS water inventory balance once every 24 hours[1, 2]

B. Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG once every 72 hours, with the following exception. The primary to secondary LEAKAGE for the Unit 1 B steam generator will be verified to be ≤ 20 gallons per day during Operating Cycle 23.[1]

Notes:

1. Not required to be completed until 12 hours after establishment of steady state operation.
2. Not applicable to primary to secondary LEAKAGE.

BASES

SURVEILLANCE REQUIREMENTS (SR)

SR 4.13.A

Verifying RCS LEAKAGE to be within the Limiting Condition for Operation (LCO) limits ensures the integrity of the reactor coolant pressure boundary (RCPB) is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions (stable pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The surveillance is modified by two notes. Note 1 states that this SR is not required to be completed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in the TS 3.1.C Bases.

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 24 hour frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 4.13.B

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG, with the following exception. The primary to secondary LEAKAGE for the Unit 1 B steam generator will be limited to 20 gallons per day during Operating Cycle 23. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.1.H, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 4. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG.

If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG; for Unit 1 that leakage should be assumed to be through the B steam generator for Operating Cycle 23. The surveillance is modified by a Note, which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The surveillance frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 4).

REFERENCES

1. UFSAR, Chapter 4, Surry Units 1 and 2.
2. UFSAR, Chapter 14, Surry Units 1 and 2.
3. NEI 97-06, "Steam Generator Program Guidelines."
4. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

components exceeds 94 degrees, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.

b. For Unit 1 Refueling Outage 22 and the subsequent operating cycle, tubes with flaws having a circumferential component less than or equal to 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet do not require plugging. Tubes with flaws having a circumferential component greater than 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet shall be removed from service.

   Tubes with service-induced flaws located within the region from the top of the tubesheet to 17 inches below the top of the tubesheet shall be removed from service. Tubes with service-induced axial cracks found in the portion of the tube below 17 inches from the top of the tubesheet do not require plugging.

   When more than one flaw with circumferential components is found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet with the total of the circumferential components greater than 203 degrees and an axial separation distance of less than 1 inch, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.

   When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet, and the total of these circumferential components exceeds 94 degrees, then the tube shall be removed from service. When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet and within 1 inch axial separation distance of a flaw above 1 inch from the bottom of the tubesheet, and the total of these circumferential components exceeds 94 degrees, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.

c. For Unit 1 Refueling Outage 22 and the subsequent operating cycle, tubes in the B steam generator with permeability variation indications that may mask flaws in the bottom one inch of the tubesheet do not require plugging.

j. Following completion of a Unit 2 inspection performed in Refueling Outage 21 (and any inspections performed in the subsequent operating cycle), the primary to secondary LEAKAGE rate observed in each steam generator (if it is not practical to assign leakage to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one steam generator) during the cycle preceding the inspection which is the subject of the report, and

k. Following completion of a Unit 2 inspection performed in Refueling Outage 21 (and any inspections performed in the subsequent operating cycle), the calculated accident leakage rate from the portion of the tube below 17 inches below the top of the tubesheet for the most limiting accident in the most limiting steam generator.

l. Following completion of a Unit 1 inspection performed in Refueling Outage 22 (and any inspections performed in the subsequent operating cycle), the number of indications and location, size, orientation, whether initiated on primary or secondary side for each service-induced flaw within the thickness of the tubesheet, and the total of the circumferential components and any circumferential overlap below 17 inches from the top of the tubesheet as determined in accordance with TS 6.4.Q.3,

m. Following completion of a Unit 1 inspection performed in Refueling Outage 22 (and any inspections performed in the subsequent operating cycle), the primary to secondary LEAKAGE rate observed in each steam generator (if it is not practical to assign leakage to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one steam generator) during the cycle preceding the inspection which is the subject of the report,

n. Following completion of a Unit 1 inspection performed in Refueling Outage 22 (and any inspections performed in the subsequent operating cycle), the calculated accident leakage rate from the portion of the tube 17 inches below the top of the tubesheet for the most limiting accident in the most limiting steam generator, and

o. Following completion of a Unit 1 inspection performed in Refueling Outage 22 (and any other inspections performed in the subsequent operating cycle), for the B steam generator, the number of permeability variation indications including location and total circumferential extent.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 264 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NO. 1

DOCKET NO. 50-280

1.0 INTRODUCTION

By letter dated May 5, 2009 (Agencywide Documents Access and Management System (ADAMS) accession no. ML091270444), Virginia Electric and Power Company (the licensee) submitted an emergency license amendment request to change the technical specifications (TS) for Surry Power Station, Unit 1 (SPS Unit 1). The licensee provided additional information in a letter dated May 6, 2009. The request proposed changing the reactor coolant system (RCS) operational leakage limits of TS Section 3.1.C, "RCS Operational Leakage," and TS Section 4.13 "RCS Operational Leakage," to 20 gallons per day (gpd). The request also proposed changing the repair requirements of TS Section 6.4.Q, "Steam Generator (SG) Program," and to the reporting requirements of TS Section 6.6.A.3, "Steam Generator (SG) Tube Inspection Report." The proposed changes would establish alternate repair criteria for SG B of Unit 1, which would allow tubes with a permeability variation in the lowest one inch of the tubesheet to remain in service during Refueling Outage 22 (spring 2009) and the subsequent operating cycle. The licensee also included changes to the bases discussions in TS Bases Section 3.1.C "RCS Operational Leakage," and TS4.13 "Surveillance Requirements," for information. Additionally, the licensee made a regulatory commitment to include a leakage factor ratio of 4.7 between calculated accident leakage and normal operating leakage.

The supplemental letter dated May 6, 2009, provided additional information that clarified the application, but did not expand the scope of the original proposed no significant hazards consideration determination as appeared in the original May 5, 2009, application.

In its letter dated May 5, 2009, the licensee submitted Westinghouse Electric Company's (WEC) letter, LTR-SGMP-09-63, "Assessment of Tube-end Indications at Surry Unit 1," dated May 4, 2009, as a technical basis to support the emergency license amendment request of the plant's TSs.

In its supplemental letter dated May 6, 2009, the licensee submitted WEC letter, LTR-SGMP-09-64 Rev. 2 P-Attachment, "Surry Unit 1 Support for Emergency Technical Specification Amendment to Address Permeability Variation Signals at the Tube Ends," dated May 6, 2009. Because the letter contained proprietary information, the submission included

Affidavits, signed by the licensee, requesting that the NRC withhold the proprietary information from the public.

NRC letters regarding the withholding of proprietary information in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), paragraph 2.390(b)(5) and Section 103(b) of the Atomic Energy Act of 1954, as amended, will be issued. There is no proprietary information in this safety evaluation (SE).

2.0 REGULATORY EVALUATION

In Title 10 of the *Code of Federal* Regulations (10 CFR), Part 50, Section 50.36, the Commission established its regulatory requirements related to the content of the TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TSs. In 10 CFR 50.36(d)(5), administrative controls are stated to be, "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner." This also includes the programs established by the licensee and listed in the administrative controls section of the TS for the licensee to operate the facility in a safe manner. For SPS Unit 1, the requirements for performing SG tube inspections and repair are in TS 3.1.H and TS 6.4.Q, while the requirements for reporting the SG tube inspections and repair are in TS 6.6.A.3.

The TSs for all pressurized water reactor (PWR) plants require that an SG program be established and implemented to ensure that SG tube integrity is maintained. For SPS Unit 1, SG tube integrity is maintained by meeting the performance criteria in TS 6.4.Q.2, for structural and leakage integrity, consistent with the plant design and licensing basis. TS 6.4.Q.1 requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected, to confirm that the performance criteria are being met. TS 6.4.Q also includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspections be performed with the objective of detecting flaws of any type that may be present in the entire tube and that may satisfy the applicable tube repair criteria, except as indicated in this amendment request regarding the application of a limited inspection scope in the tube sheet region. The applicable tube repair criteria, specified in TS 6.4.Q.3, are that tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged, except if permitted to remain in service through application of the alternate repair criteria provided in TS 6.4.Q.3.

The SG tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and isolate fission products in the primary coolant from both the secondary coolant and the environment. For the purposes of this SE, SG tube integrity means that the tubes are capable of performing these safety functions in accordance with the plant design and licensing basis.

The General Design Criteria (GDC) in Appendix A of 10 CFR Part 50 provide regulatory requirements which state that the RCPB shall have, "an extremely low probability of abnormal leakage...and gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDCs 15 and 31), shall be of, "the highest quality standards practical" (GDC 30), and shall be designed to permit, "periodic inspection and testing...to assess...structural and leaktight integrity" (GDC 32). To this

end, 10 CFR 50.55a specifies that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code* (ASME Code). Section 50.55a further requires that throughout the service life of a PWR facility, like SPS Unit 1, ASME Code Class 1 components meet the requirements of the ASME Code to the extent practical, except for design and access provisions; and pre-service examination requirements in Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." This requirement includes the inspection and repair criteria of Section XI of the ASME Code. The Section XI requirements pertaining to inservice inspection of SG tubing are augmented by additional requirements in the TSs.

As part of the plant licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents (DBAs) such as a SG tube rupture and a main steam line break (MSLB). These analyses consider primary-to-secondary leakage that may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits for offsite doses, or for control room operator doses (or some fraction thereof as appropriate to the accident), or the NRC-approved licensing basis (e.g., a small fraction of these limits). No accident analysis for SPS Unit 1 is being changed because of the proposed amendment and, thus, no radiological consequences of any accident analysis are being changed. The licensee-proposed changes to TS 6.4.Q stay within the GDC requirements for the SG tubes and maintain the accident analysis and consequences that the Nuclear Regulatory Commission (NRC) has reviewed and approved for the postulated DBAs for SG tubes.

The proposed amendment is applicable to Refueling Outage 22 and the subsequent operating cycle of SPS Unit 1. This license amendment differs from the previously approved interim alternate repair criteria amendment by allowing tubes in SG B that have permeability variations in the bottom one inch to remain in service. As a compensating measure for this amendment, the allowable operating leakage limit for SG B is being lowered from 150 gpd to 20 gpd, to ensure a more timely response to any detected leakage in SG B. An additional reporting requirement is also being added to TS 6.6.A.3.

3.0 TECHNICAL EVALUATION

3.1 Proposed Changes to the TSs

TS 3.1.C – RCS Operational Leakage

TS 3.1.C.1.d is being revised to limit the reactor coolant system (RCS) primary to secondary leakage for the Unit 1 B SG to 20 gallons per day during the Operating Cycle 23:

> *d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG), with the following exception. The primary to secondary LEAKAGE for the Unit 1 B steam generator will be limited to 20 gallons per day during Operating Cycle 23.*

TS Bases 3.1 - RCS Operational Leakage

TS Bases 3.1, Applicable Safety Analysis and Limiting Condition for Operation, section d, Primary to Secondary LEAKAGE through Any One SG, is being revised to add a discussion on the basis

for limiting the primary to secondary leak rate to 20 gallons per day for Operating Cycle 23 on Unit 1 B SG. The following sentence will be added:

> *Due to the permeability variation indications in the Unit 1 B steam generator found during Refueling Outage 22, the primary to secondary leak rate for that steam generator is limited to 20 gallons per day for Operating Cycle 23.*

TS 4.13 - RCS Operational Leakage

TS 4.13.B is being revised to limit the RCS primary to secondary leakage for the Unit 1 B steam generator to 20 gallons per day during the Operating Cycle 23:

> *B. Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG once every 72 hours[1], with the following exception. The primary to secondary LEAKAGE for the Unit 1 B steam generator will be verified to be ≤ 20 gallons per day during Operating Cycle 23.*

TS Bases Surveillance Requirements 4.13.B

TS Bases 4.13.B is being revised to discuss the 20 gallons per day primary to secondary leak rate limit on Unit 1 B steam generator for Operating Cycle 23. The first sentence of the paragraph will be revised as follows:

> *This SR verifies that the primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG, with the following exception. The primary to secondary LEAKAGE for the Unit 1 B steam generator will be limited to 20 gallons per day during Operating Cycle 23.*

TS Bases 4.13.B is also being revised to indicate that if the leakage rate cannot be assigned to an individual generator, then all the leakage should be conservatively assigned to one SG:

> *If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG; for Unit 1 that leakage should be assumed to be through the B steam generator for Operating Cycle 23.*

TS 6.4.Q.3 relates to SG tube repair criteria and 6.4.Q.3.c. is being added as shown below:

> *c. For Unit 1 Refueling Outage 22 and the subsequent operating cycle, tubes in the B steam generator with permeability variation indications that may mask flaws in the bottom one inch of the tubesheet do not require plugging.*

TS 6.6.A.3 contains the reporting requirements for the SG Tube Inspection Report and 6.6.A.3.o. is being added as shown below:

> *o. Following completion of a Unit 1 inspection performed in Refueling Outage 22 (and any other inspections performed in the subsequent operating cycle), for the B steam generator, the number of permeability variation indications including location and total circumferential extent.*

Due to the addition of TS 6.6.A.3.o., the word "and" is being deleted from the end of TS 6.6.A.3.m. and being added to the end of TS 6.6.A.3.n., to maintain proper syntax.

3.2 Proposed Change to Commitment Regarding Leakage Factor to be used in Condition Monitoring and Operational Assessments

As part of the Unit 1 IARC amendment, a regulatory commitment was made by the licensee to apply a leakage factor of 2.5 to the leakage observed during normal operation that cannot be associated with the degradation mechanisms outside the tube sheet expansion region. For integrity assessments, the leakage factor of 2.5 was used for completion of both the condition monitoring (CM) and operational assessment (OA), upon implementation of the IARC. For example, for the CM assessment, the component of leakage from the lower 4 inches for the most-limiting steam generator during the prior cycle of operation would have been multiplied by a factor of 2.5 and added to the total leakage from any other source and compared to the allowable accident analysis leakage assumption. For the OA, the difference in leakage between the allowable limit, during the limiting design basis accident, and the leakage from other sources would have been divided by 2.5 and compared to the observed operational leakage.

The licensee has again committed to using the leakage factor methodology; however, additional margin has been provided by increasing the leakage factor from 2.5 to 4.7. The leakage factor of 4.7 will be used in the CM and OA integrity assessments as described above.

3.3 Technical Evaluation

The IARC amendment for SPS Unit 1 issued on April 8, 2009, is applicable only to Refueling Outage 22 (spring 2009) and the subsequent operating cycle. The IARC amendment is intended by the licensee to be a conservative interim requirement pending a future license amendment incorporating an "H*" methodology.

The thesis of the H* approach is that the entire end cap load (including margins consistent with the design basis) is reacted out of the tube by tube-to-tubesheet friction at some distance below the top of the tubesheet, which is defined as the H* distance and which is less than the thickness of the tubesheet.

The staff has reviewed license amendment requests for a number of plants over the past several years. These amendment requests were either withdrawn or rejected by the staff in view of a number of unresolved issues identified by the staff (ADAMS accession no. ML080450185). The H* methodology has been substantially revised since that time to address the issues raised by the staff. The staff has had numerous meetings with Westinghouse and interested licensees to discuss the revised approach (ADAMS accession nos. ML083300571, ML083540708, ML090080195, ML090370928, and ML091210437). A key element of the Westinghouse analysis involves a 3-D finite element model of the lower SG assembly including the lower SG shell, tubesheet, lower SG channel head, and SG divider plate for purposes of predicting tubesheet displacements at tube bore hole locations. An NRC contractor, Argonne National Laboratory (ANL), independently developed its own finite element model as part of NRC sponsored research. ANL was successful in reconciling tubesheet displacement predictions with those by Westinghouse.

Using its revised approach, Westinghouse has completed H* calculations for Model F SGs, indicating an H* distance of approximately 12-inches as compared to the 21- inch thickness of the tubesheet. It is the Westinghouse position that the pressure end cap loads on the tubes are totally reacted out of the tube over the H* distance and that the portion of the tube below this distance transmits zero load. Thus, Westinghouse concludes that cracks below the H* distance have no affect on tube pullout resistance.

Westinghouse has not completed similar calculations for Model 51 SGs, including SPS Unit 1. In its letter dated May 6, 2009, the licensee stated that based on an assessment of differences between the SPS Unit 1 Model 51 SGs and the Model F SGs, Westinghouse estimates that the H* calculations for SPS Unit 1 will show an H* value of about 12.5 inches, when complete, compared to 12-inches for Model F.

Although the H* calculations for SPS Unit 1 have not been completed nor formally reviewed by the NRC staff, the staff has not identified any issues in its interactions with Westinghouse and interested licensees over the past year which call into question the viability of defining a conservative H* value which is less than the thickness of the tubesheet for the various model SGs. Thus, although the staff has not reached conclusions about specific values of H* that are acceptable for the different SG models, the staff finds that the available information supports a conclusion that the lowermost 1-inch of tubing is outside the load path for the pressure end cap loads. Thus, even if the tubes are assumed to be totally severed at the lowermost 1-inch of the tube, the staff concludes that slippage of the tubes is not expected under the full range of normal operating, transient, and accident conditions.

The licensee's May 6, 2009, letter includes an assessment of hypothetical slippage of the tubes under accident conditions assuming all the tubes with permeability indications are severed at the bottom of the tubesheet and there is zero friction between the tubes and tubesheet. These analyses show that the maximum hypothetical slippage is limited by intact tubes and SG support structures to 8.7 inches, which is significantly less than the 21-inch tubesheet thickness. Thus, the tubes are constrained against total tube pullout condition (i.e., gross structural failure).

Based on its conclusion that no slippage of the tubes is expected, even if the tubes are totally severed at the bottom of the tubesheet, the 2.5 leakage factor (accident-induced leak rate divided by normal operating leak rate) that the licensee committed to as part of the April 8, 2009, IARC amendment remains conservative. As part of the current license amendment request, the licensee has committed to a more conservative leakage factor of 4.7 to ensure that calculated accident-induced leak rate will continue to be conservative.

Industry-wide, within the U.S., there have been no reported occurrences of observed primary to secondary leakage from cracks located within the thickness of the tubesheets where alloy 600 thermally treated tubes have been hydraulically expanded against the tubesheets. Given the cracks reported in the tubes within the tubesheet thickness at a number such plants, including those in SG A and C of SPS Unit 1, the absence of reported leakage from the tubesheet region suggests that the tight tube-to-tubesheet crevices provide significant leak resistance. SPS Unit 1 has operated with negligible primary to secondary leakage (< 1 gpd, with no leaking tubes identified during subsequent inspections), from any source inside or outside of the tubesheets, over the last several cycles. Thus, significant leakage during the upcoming cycle is not expected. However, the licensee has included as part of the proposed amendment, a change to the operational primary to secondary leak rate limit in TS 4.13. This change would incorporate the

following exception to current value of 150 gpd through any one SG. This exception states, "The primary to secondary LEAKAGE for the Unit 1 B steam generator will be limited to 20 gallons per day during Operating Cycle 23." Should there be leakage from SG B during the upcoming cycle, the reduced leak rate limit provides added assurance that any unexpected conditions developing in the tubesheet region of SG B will be detected and corrected in a timely manner.

4.0 SUMMARY

Based on the above evaluation, the staff finds that the proposed license amendment, which is applicable only to Refueling Outage 22 and the subsequent operating cycle of SPS Unit 1, ensures that SG tube structural and leakage integrity will be maintained during this period. Structural safety margins that are consistent with the design basis, and leakage integrity that is within the assumptions employed in the licensing basis accident analyses, will also be maintained. Additionally, there will be no adverse impact on the ability of the tube-to-tubesheet welds to perform their safety-related function. Based on these findings, the staff further concludes that the proposed amendment meets 10 CFR 50.36 and, thus, the proposed amendment is acceptable.

5.0 EMERGENCY CIRCUMSTANCES

The NRC's regulations 10 CFR 50.91 contain provisions for issuance of an amendment where the Commission finds that emergency circumstances exist, in that a licensee and the Commission must act quickly and that time does not permit the Commission to publish a *Federal Register* notice allowing 30 days for prior public comment.

In the May 4, 2009, application, the licensee requested that this amendment be treated as an emergency amendment. In accordance with 10 CFR 50.91(a)(5), the licensee provided information explaining how this emergency situation occurred and why it could not be avoided.

5.1 Reason Emergency Situation Has Occurred:

On May 1, 2009 at approximately 10:10 a.m., NRC received a call from the licensee that an emergency situation exists, that failure to act would result in a delay in resumption of operation or of an increase in power output up to the plant's licensed power level. The B SG tube inspection commenced on April 30, 2009, at 12:06 p.m. The Unit 1 IARC was approved by the NRC on April 8, 2009, by TS Amendment 263. Based on the Unit 1 IARC requirements, a first time inspection of the B SG tube ends was performed using a rotating eddy current probe. The B SG inspection identified a large number of permeability variation indications of significant signal strength requiring disposition. These indications are unique to the Unit 1 B SG. The locations were reexamined using additional diagnostic techniques. Examinations included the use of magnetic bias probes, often useful in distinguishing permeability variations from flaws. These examinations were not successful due to the lack of significant reduction in the voltage signal with the magnetic bias probe associated with the geometry of the location and the strength of the permeability signal. In summary, the use of additional diagnostic techniques did not resolve the masking issue. Preliminary disposition of the permeability variation indications concluded that these indications may mask flaws in the B SG tubes. A Condition Report in the Corrective Action Program was issued on May 1, 2009, documenting the permeability variation indications and the potential masking of flaws. The inspection was completed on May 3, 2009, at 06:10 a.m. The 200 °F mode change for startup from Unit 1 Refueling Outage 22 is currently scheduled to commence at 04:00 a.m. on May 8, 2009.

To prevent delay in either resumption of operation or of increase in power output up to the plant's licensed power level of Surry Unit 1 the modified IARC for the Unit 1 B SG is needed to support the increase in RCS temperature above 200 °F.

5.2 Reason the Situation Could Not Have Been Avoided:

The Unit 1 IARC was approved by the NRC on April 8, 2009, by TS Amendment 263 which was implemented on April 29, 2009. The SG tube inspection scope for Unit 1 Refueling Outage 22 included a planned inspection in the A and C SGs, as well as a pre-planned scope expansion for the B SG, if required. As a result of tube end cracks found in the A and C SGs, the pre-planned expansion of the SG tube inspection in the B SG was undertaken. The B SG tube inspection commenced on April 30, 2009, at 12:06 p.m., and was completed on May 3, 2009, at 06:10 a.m. Based on the Unit 1 IARC requirements, a first time inspection of the B SG tube ends was performed using a rotating eddy current probe. The B SG inspection identified a large number of permeability variation indications of significant signal strength requiring disposition. This would require plugging of SG tubes in the Unit 1 B SG with permeability variation indications in the bottom one inch of the tubesheet unless the TS are changed as discussed above.

Based on the above discussion, the licensee has submitted a letter dated October 14, 2008, and requested a license amendment for SPS Unit 1 to incorporate an IARC into the provisions for SG tube repair for use during SPS Unit 1 Refueling Outage 22 and the subsequent operating cycle. A February 20, 2009, letter provided the response to an NRC request for additional information, including a commitment to use the 2.5 leakage factor. The Unit 1 IARC was approved by the NRC on April 8, 2009, by TS Amendment 263, which was implemented on April 29, 2009. Inspection was completed on May 3, 2009. Neither a routine nor an exigent TS amendment request could have been processed within the 72-hour period. Therefore, an emergency TS amendment is required to prevent delay in either resumption of operation or of increase in power output up to the plant's licensed power level of Surry Unit 1.

Additional Actions

Based on the above, the Commission finds that an emergency situation exists, in that failure to act in a timely way would result in a delay in resumption of operation or of increase in power output up to the plant's licensed power level. The licensee has explained why the emergency situation occurred and why it could not be avoided. Accordingly, the Commission has determined that emergency circumstances exist pursuant to 10 CFR 50.91(a)(5) and could not have been avoided, that the submittal of information by the licensee was timely, and the licensee did not create the emergency condition. Therefore, this request was handled under the provisions of 10 CFR 50.91(a)(5).

6.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92(c) state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or,

(2) Create the possibility of a new or different kind of accident from any previously evaluated; or,

(3) Involve a significant reduction in a margin of safety.

The following analysis was provided by the licensee in their letter dated May 5, 2009.

(1) Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

Of the various accidents previously evaluated, the proposed change only affects the steam generator tube rupture (SGTR), steam line break (SLB), and locked rotor evaluations. Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this amendment request. Another faulted load consideration is a safe shutdown earthquake (SSE); however, the seismic analysis of Model 51F steam generators has shown that axial loading of the tubes is negligible during an SSE.

At normal operating pressures, leakage from PWSCC below 17 inches from the TTS is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region.

For the SGTR event, the required structural margins of the steam generator tubes with circumferential flaws of any extent within the bottom one inch of the tubes are maintained by ensuring that the available axial force due to tube to tubesheet contact pressure and the coefficient of friction between the tube and the tubesheet exceeds the maximum pullout force for the limiting operating conditions. These forces were calculated and the results documented in Attachment 4 [Reference 7 [of the application]]. In all cases, the available axial forces are sufficient to offset tube pullout forces, thus maintaining tube integrity. Attachment 4 [Reference 7 [of the application]] further concludes that the performance criteria of NEI 97-06, Rev. 2, "Steam Generator Program Guidelines" and Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" continue to be met. Therefore, the margin against tube burst/pullout is maintained during normal and postulated accident conditions and the proposed change does not result in a significant increase in the probability or consequence of a SGTR.

The probability of a SLB is unaffected by the potential failure of a SG tube as the failure of a tube is not an initiator for a SLB event. SLB leakage is limited by leakage flow restrictions resulting from the leakage path above potential cracks through the tube–to-tubesheet crevice. The leak rate during postulated accident conditions (including locked rotor) has been shown to remain within the accident analysis assumptions for circumferential flaws of any extent located in the bottom one inch of the tubes. Consistent with the 4.7 leakage factor, the primary to secondary leakage will be administratively limited to 100 gpd, the attendant accident condition leak rate, assuming all leakage to be from indications below 17 inches from the top of the tubesheet would be bounded by 470 gpd. This value is within

the accident analysis assumptions for the limiting design basis accident for Surry, which is the postulated SLB event.

The reduction in the primary to secondary leakage limit from 150 gallons per day to 20 gallons per day for the Unit 1 B Steam Generator provides additional margin.

Based on the above, the performance criteria of NEI-97-06, Rev. 2 and Regulatory Guide 1.121 continue to be met and the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Does the proposed change create the possibility of a new or different accident from any accident previously evaluated?

Response: No

The proposed change does not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity is expected to be maintained for all plant conditions upon implementation of the modified interim alternate repair criteria. The proposed change does not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

The reduction in the primary to secondary leakage limit from 150 gallons per day to 20 gallons per day for the Unit 1 B Steam Generator can not initiate an accident.

Therefore, based on the above evaluation, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed change involve a significant reduction in a margin of safety?

Response: No

The proposed change maintains the required structural margins of the steam generator tubes for both normal and accident conditions. NEI 97-06, Rev. 2 and RG 1.121 are used as the basis in the development of the limited tubesheet inspection depth methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC staff for meeting GDC 14, 15, 31, and 32 by reducing the probability and consequences of an SGTR. RG 1.121 concludes that by determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service or repaired, the probability and consequences of a SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the ASME Code.

For circumferential flaws of any extent in the bottom one inch of the tubes Attachment 4 [Reference 7 [of the application]] documents that tube to tubesheet contact pressure combined with a conservative coefficient or friction serve to provide the necessary resistance to tube pullout due to the pressure and thermally induced forces (with applicable safety factors applied). Additionally, it is shown that the existence of

circumferential indications of any extent located in the bottom one inch of the tubes will not result in unacceptable primary to secondary leakage during all plant conditions.

The reduction in the primary to secondary leakage limit from 150 gallons per day to 20 gallons per day for the Unit 1 B Steam Generator does not result in a reduction in margin since operator intervention will be required at the lower leakage limit.

Based on the above, it is concluded that the proposed change does not result in any reduction of margin with respect to plant safety as defined in the Updated Final Safety Analysis Report or bases of the plant Technical Specifications.

The NRC staff agrees with the licensee's analysis and, based on this review has concluded that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff makes a final determination that the amendment does not involve a significant hazards consideration.

7.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendment. The State official had no comments.

8.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes the surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final no significant hazards finding with respect to this amendment. The amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

9.0 CONCLUSION

The NRC staff has concluded, based on the considerations discussed above, that (1) the amendment does not (a) involve a significant increase in the probability or consequences of an accident previously evaluated or, (b) create the possibility of a new or different kind of accident from any previously evaluated or, (c) involve a significant reduction in a margin of safety and therefore, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (3) such activities will be conducted in compliance with the Commission's regulations, and (4) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: A. Johnson, DCI/CSGB

Date: May 7, 2009

May 7, 2009

Mr. David A. Christian
Senior Vice President
and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION, UNIT NO. 1, ISSUANCE OF AMENDMENT REGARDING MODIFIED INTERIM ALTERNATE REPAIR CRITERIA FOR B STEAM GENERATOR TUBE REPAIR (TAC NO. ME1191)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 264 to Renewed Facility Operating License No. DPR-32 for the Surry Power Station, Unit No. 1. The amendment changes the Technical Specifications (TSs) in response to your application dated May 5, 2009, as supplemented by letter dated May 6, 2009.

The proposed amendment modifies the Unit 1 interim alternate repair criteria (IARC) in TS 6.4.Q, "Steam Generator (SG) Program," and TS 6.6.3, "Steam Generator Tube Inspection Report." The proposed emergency license amendment request incorporates a modified IARC for the Unit 1 B SG tube repair criteria during Unit 1 Refueling Outage 22 and the subsequent operating cycle and revises the reporting requirements for the Unit 1 B SG.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA DWright for/

John Stang, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-280

Enclosures:
1. Amendment No. 264 to DPR-32
2. Safety Evaluation

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Amendment No.: ML091260386 *SE transmitted by letter dated

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DATE	05/07/09	05/07/09	05/07/09	05/07/09
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NAME	MGavrilas	SRohrer for MO'Brien	AJones	RMartin for MWong
DATE	05/6/09	05/07/09	05/7/09	05/07/09

*** comments addressed** **OFFICIAL RECORD COPY**